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ABSTRACT

As part of a continuing effort by the Office of Fusion Energy to define an ignition experiment, a superconducting tokamak has been designed with thin neutron shielding and aggressive magnet and plasma parameters. By so minimizing the inner radial dimensions of the tokamak center post, coil, and shielding region, the plasma major radius is reduced with a corresponding reduction in device costs. The peak nuclear-heating rate in the superconducting TF coils is 22 mW/cm^3, which results in a steady heat load to the cryogenic system of 50 kW. Fast-wave, lower-hybrid heating would be used to induce a 10-MA current in a moderate density plasma. Then pellet fueling would raise the density to achieve ignition as the current decays in a few hundred seconds. Steady-state current drive in subignited conditions permits a 0.8 MW/m^2 average wall loading to study plasma and nuclear engineering effects.

INTRODUCTION

A Tokamak Ignition/Burn Experimental Research device (TIBER) has been designed to represent the smallest superconducting ignition experiment consistent with the OFE Mission II physics and engineering objectives. To reduce the size and cost of the tokamak, more aggressive design assumptions were necessary. Plasma shaping is used to achieve higher beta; and neutron shielding is minimized. However, the superconducting TF magnets must be sufficiently shielded to reduce neutron heat load to the coolant, neutron fluence to the superconductor and stabilizer, and gamma doses to the insulator.[1] In particular, the insulation must retain adequate strength and electrical properties after irradiation to end-of-life fluences above 10^{19} n/cm^2 and gamma dosages above 10^{10} rads, especially in those portions of the magnet adjacent to shield penetrations for diagnostics on plasma-heating systems. The peak nuclear heating rate in the superconducting TF coil is 22 mW/cm^3, resulting in a total system heat load of 50 kW. Both the TF and PF coils would be constructed of force-cooled Nb_3Sn to achieve high-current density with acceptable nuclear heat removal and neutron damage limits.

DESIGN DESCRIPTION

A point design of the TIBER device is summarized in Table I. As depicted in Fig. 1, a high-field (14-T) pusher coil is used in the normal position for the ohmic heating coil. By effecting a modest plasma indentation, it achieves a beta of 10 percent. All inner-leg components of the tokamak must be kept as small as possible so that the plasma radius can be minimized and plasma shaping maximized. Accordingly, magnet current densities of 4 kA/cm^2 and an integrated structural design are necessary. The neutron shielding of the inner leg ranges between 40 and 45 cm.

Table I. TIBER point design parameters.

Major plasma radius (m)	2.60
Minor plasma radius (m)	0.73
Elongation	1.94
Triangularity	0.60
Indentation	0.05
Average beta (%)	10
Safety factor	3.80
Plasma current (A)	6.6×10^6
TF coil (Nb_3Sn) at 4.5 K (T)	10
PF coil (Nb_3Sn/NbTi) at 1.8 K (T)	14
Peak nuclear heating in superconducting coil (mW/cm^3)	22
Fusion power (MW)	194
Current drive power (MW)	19
Plasma Q	10

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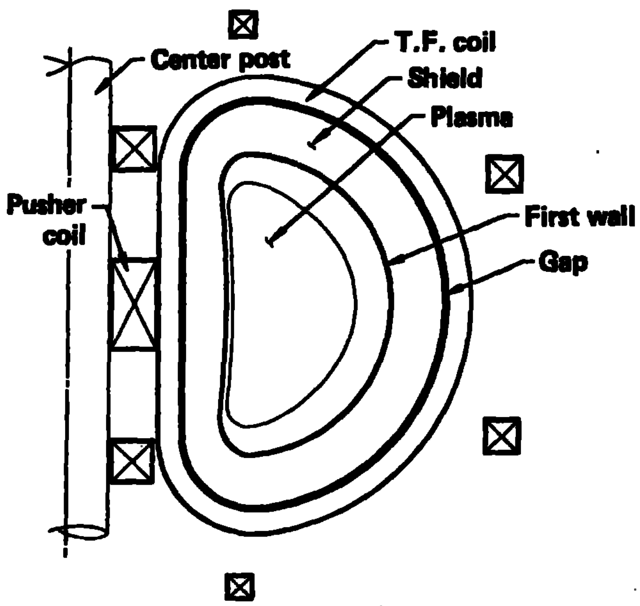


Fig. 1. TIBER cross section with pusher coil and shaped plasma.

Noninterlinking TF and PF coils are used in TIBER for easy maintenance. Also, the entire tokamak is enclosed in a single vacuum vessel, similar to the practice on the Mirror Fusion Test Facility. In this way internal dimensions and complications are minimized, leading to easily serviced external vacuum joints for rapid disassembly.

The critical part of the TIBER configuration is the region between the center post and the plasma at the midplane of the machine. The radial build of the post coils and shield determines the overall size of the device. Accordingly, we have concentrated on the structural design of the pusher and TF coils in the area of the pusher coil. The magnetic pressure on the inboard side of the TF coil case is about 60 MPa (9000 psi), which is partially offset by the radial outward force on the pusher coil, which may develop as much as 20 MPa (3000 psi) [no credit is taken for this in the design]. There is also a large hoop stress on the TF coil which will be shared by the TF coil case and the center post.

All of the structural elements will be operating at 1.8 to 4.3 K, which will allow the use of high design stresses. We have selected a peak design stress limit of 536 MPa (80,000 psi) for the stainless steel structure of the pusher coil and those portions of the TF coil in the region of the pusher coil. The critical region of the TF coil has been modeled with the stress analysis code GEMINI, which has proved to be a useful tool to help us understand the complex stresses and deflections in the coil.

To conserve space in TIBER, we have considered both a pumped limiter and an open, internal divertor. Either choice requires an analysis of the edge plasma and recycling at the dump plates. Initial calculations indicate that a large recycling fraction is needed to cool the edge plasma and reduce the sputter erosion at the dump plates and at the walls. The choice of parameters and of either a pumped limiter or an open divertor depends on the space available outside the separatrix. This requires a compromise between the space available for neutron shielding of the magnets and the space needed for recycling the edge plasma.

The dump plates in TIBER are similar to those in the halo dump in MARS.[2] The plates are sized and contoured to handle the heat load without regard to gas removal. The small pumping ports that remove the gas through the plates are located as needed to control the amount of recycling. In the case of a pumped limiter, this vented-port concept has the additional advantage of moving the leading edge out of the plasma to where heat and erosion problems are minimized. Figure 2 shows a sketch of a pumped limiter using this concept. Each small port has its own leading edge, but the large area surrounding allows adequate cooling.

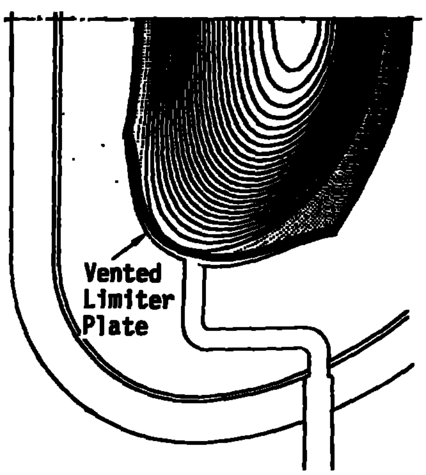


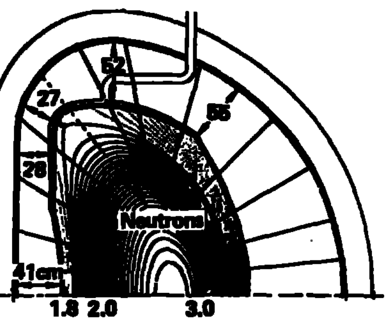
Fig. 2. A pumped limiter with a ported neutralizer plate. Although not clear from the figure, the field lines intersect the plate at grazing angles of incidence.

Because of the toroidal magnetic field, the plasma contacts the dump plates at grazing incidence and cannot stream through the small ports to be removed by the vacuum pumps. Gas is removed through the ports when the gas pressure on the plasma side of the plates exceeds that in the pumping duct behind the plates. This sets an upper limit on the gas pressure in the duct, but this limit is rather high. When the recycling fraction of the edge plasma is sufficiently high to protect the dump plates, the gas pressure on the plasma side must be nearly equal to the plasma pressure there; this pressure is on the order of 0.1 Torr. It should, therefore, be possible to obtain a pressure of a few tens of millitorr in the duct, which would allow mechanical pumps to remove the gas. The duct space behind the plates can be thin, tapering up to only 4 cm maximum, because of the high pressure and because the duct space extends completely around the torus.

One 125-mm-diameter vacuum pipe at the top and one at the bottom passes between each of the TF coils. The shielding displaced by these pipes is located between the coils, so that the neutron flux to the coils is only increased slightly.

NUCLEAR HEATING OF TF COILS

Estimates of nuclear heating in the TF coils for 3 cases are given in Fig. 3 in which the major radius and/or the mode of plasma removal (divertor or pumped limiter) are varied. These estimates are preliminary and are based on 1-D slab-shielding calculations applied to the TIBER geometry using a generic tokamak-neutron-source distribution. There is significant uncertainty in these estimates. The major element in the shield is tungsten, making it space efficient, but expensive. Thus, it should be used only where space is limited, mainly on the inner leg.



Case	P-fusion (mW)	Heating (kW	Wall loading (mW/m²)
1: R = 2.6 with divertor (shown)	200	415	2.1
2: R = 2.73 m with divertor	237	96	2.3
3: R = 2.6 m with pumped limiter	200	43	2.1

*Estimated based on 1-D approximation with W shielding and a generic neutron source distribution

Fig. 3. TIBER—estimate of nuclear heating in the coils and cases and in the first wall loading at midplane of outer leg.

Using this generic neutron source distribution from Ref. 3 results in a peak neutron wall loading greater than 2 MW/m^2 on the outer leg. This distribution is very favorable because it allows for the possibility of a nuclear component test program at near reactor levels, even though the average wall loading is less than 1 MW/m^2. From the FINESSE Study certain groups of tests suitable for TIBER have been identified. These are:

- Structural mechanics
- Breeder/multiplier structure interaction
- Thermal hydraulics
- Tritium production
- Nuclear heating and low fluence damage
- Instrumentation and control

PLASMA OPERATION

As illustrated in Fig. 4, we envision that fast-wave, lower-hybrid heating would be used to induce a plasma current above 10 MA at low-plasma density around 0.4 x 10^{14} ions/cm^3. Then, a laser-driven pellet fuel injector would raise the density to 3 x 10^{14} ions/cm^3, with continued fast- and slow-wave, lower-hybrid heating to achieve ignition. Because the current drive is not efficient at high-plasma density, the current would decay with a time constant of 500 seconds. During this period all plasma heating would be turned off in order to study pure ignition physics. As the plasma current decays to 7 MA and density drops to 10^{14} ions/cm^3, the slow-wave, lower-hybrid heating would again be used to sustain a steady-state plasma with a wall loading of 0.8 MW/m^2 and a plasma Q of about 10. Thus, the TIBER device can be used to study ignition physics and can also be operated in a steady-state, current-driven mode to study plasma/wall interactions, helium ash removal, and numerous neutron-damage effects.

The confinement scaling used for the TIBER operation depicted in Fig. 4 is separated into different scalings for ions and electrons. The electron confinement is taken to be neoalcator, $\chi_e = 1.5\ f_e a \sqrt{\kappa}/(R^2 n_{20})$, where f_e is an anomaly factor to account for high beta deterioration ($f_e = 2$ is used). The ion confinement is taken to be neoclassical

$$\chi_i = 4.1 \times 10^{-2}\ f_i (R/a\sqrt{\kappa})^{3/2}\ q^{-2}\ n_{20}/(B_t^2\ T_{10}^{1/2})$$

where f_i is an anomaly factor ($f_i = 2$). Using the Troyon-Wesson beta limit

$$\langle\beta_t\rangle \lesssim 0.04\ \frac{I_p(MA)}{aB_t(T)}\ ,$$

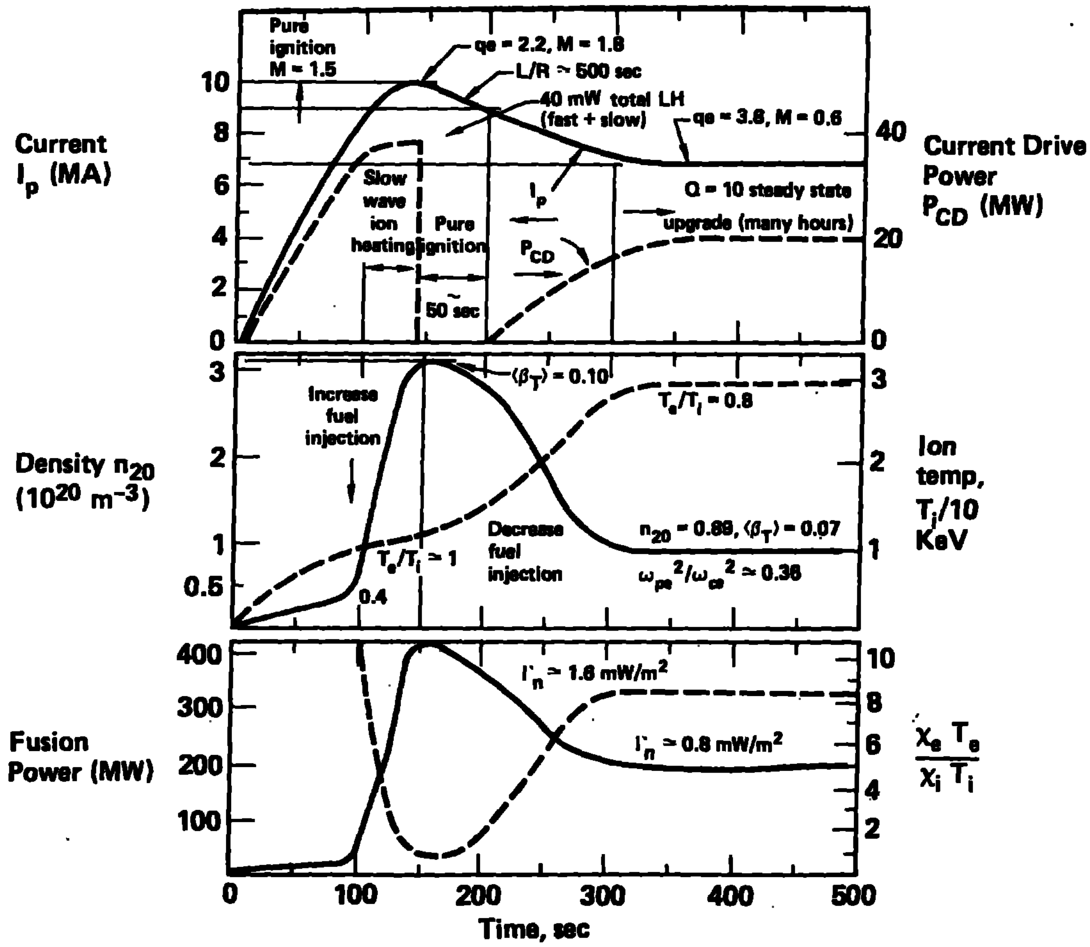


Fig. 4. Both pure ignition and high-Q steady-state physics can be explored in a Mission II device.

the ignition condition is

$$I^2(\text{MA}) \geq \frac{938\ \text{M}a\text{T}_{10}\left(1 + \dfrac{\chi_i\ T_i}{\chi_e\ T_e}\right)}{R^2\ B_t^2\sqrt{\kappa}\ f_\alpha}\ \frac{T_e}{T_i}\left(1 + \frac{T_e}{T_i}\right)^2$$

where M is the ignition margin. During the 50-s high-density, pure-ignition period in Fig. 4, M > 1.5 and $T_e/T_i \approx 1$. During the steady-state current drive (t > 300 s in Fig. 4), M = 0.6 and $T_e/T_i \approx 0.8$ result from the high temperatures with the current drive power input augmenting the alpha heating. The current-drive power input P_{cd} is calculated from Karney and Fisch[4]:

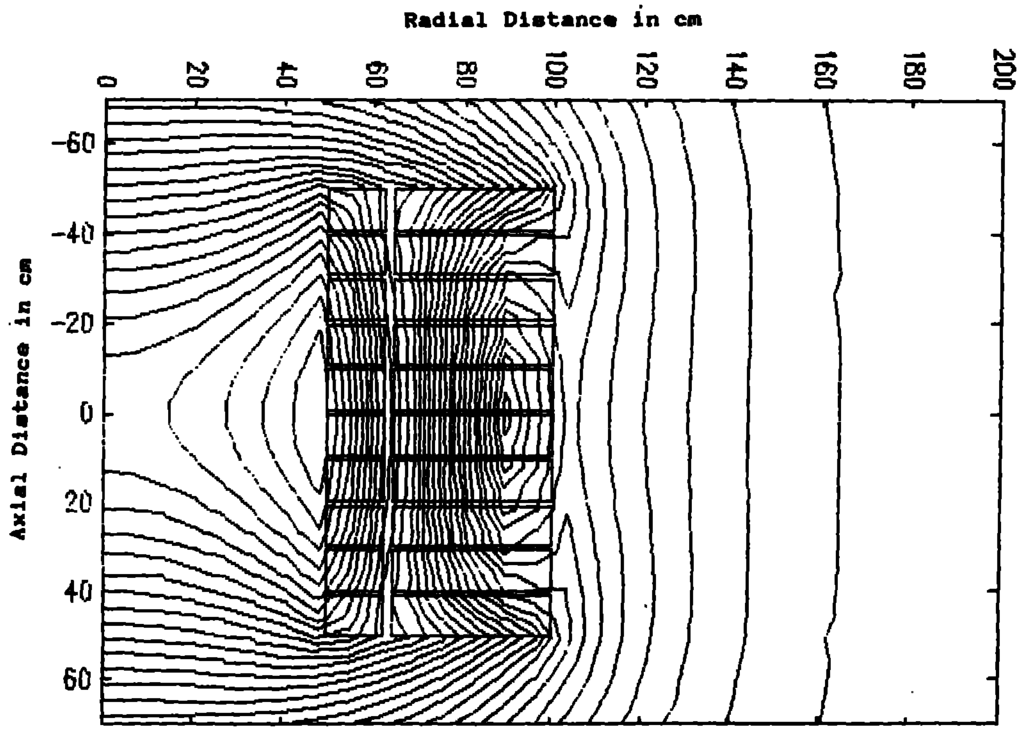
$$P_{cd} = 2.78\ \frac{I_p(\text{MA})\ R(\text{m})\ n_{20}}{\left[1 + \left(\dfrac{T_{10}}{2.5}\right)1.16\right]}\ ,$$

where the $1 + (T_{10}/2.5)^{1.16}$ factor in P_{cd} is a fit to Karney and Fisch curves for current drive efficiency, taking into account favorable relativistic effects at high T_e, and averaged over parallel indexes $\eta_{\parallel}$ = 1.5 to 2.0.

MAGNET DESIGNS

One of the more challenging magnet designs is the 14-T pusher coil shown in Fig. 5. However, we utilized conductor designs that have been proven to be both functional and within the present manufacturing capabilities. The outer section of the pusher coil is the niobium titanium used for the MFTF yin-yang coils with an inner niobium tin coil like that used for the MFTF choke coil. Each coil is immersed in superfluid helium to increase its current density and stability.

The original MFTF yin-yang conductor produced a peak field of 7.8 T at 4.2 K. Reduced temperature operation at 1.8 K yields an even higher critical current in the superconductor at 10-T fields. This implies that the current in the original MFTF winding pack can be increased by 28% to satisfy the goal of the Nb-Ti background coils. The heat flux for stability in the high-field pusher is $(1.28)^2 \times 0.19$ or 0.31 W/cm², which is manageable by He-II. Hence, the yin-yang conductor in the outer section of the pusher coil can operate



Nb_3Sn:Ti Subcoils

J_{pack}: 3700 amps/cm^2
I_o: 2500 amps
2.7 MAT
B_{max}: 14.0 T

Nb-Ti Subcoils

J_{pack}: 4000 amps/cm^2
I_o: 7392 amps
13.5 MAT
B_{max}: 10.0 T

Fig. 5. The TIBER pusher (plasma shaping) coil.

at a current of 7399 A. Hoop forces must be reacted to the external case structure.

For the inner high-field section of the pusher coil, a Nb_3Sn:Ti conductor originally tested at 12.7 T for MFTF can be used with He-II in fields of 14 T due to the low temperature enhancement of J_c. Thus we can use a conductor for the insert pusher subcoils similar to the Nb_3Sn:Ti fabricated by Furakawa for the MFTF high-field choke coils, which operates at a peak field of 12.7 T. However, in this application the extra copper stabilizer should be annealed for high-conductivity to limit the wetted perimeter heat fluxes to 1.0 W/cm^2. This implies that the large 335 MPa stresses must be reacted to prevent damage to the strain-sensitive Nb_3Sn:Ti. Since the radial pack thickness is only 12.5 cm, the coil design permits transmitting the conductor stresses to a 2-cm-thick stainless-steel outer case surrounding the Nb_3Sn:Ti subcoils. The coil consists of twenty subcoils each clad in 1-cm-thick coil cases in order to increase the effective coil modulus and transmit the radially compressive forces from the TF coils to the additional supporting structure located at the magnet bore. The fringe fields from this coil supply an approximate 0.5 T on the plasma "bean-contour".

The TF coils summarized in Table II for TIBER are made of internally cooled Nb_3Sn conductors similar to those used in the Westinghouse LCP coil and the MIT test coil. The tensile load on the straight leg of the TF coil has been calculated. Since the conductor loads are transmitted first to the sheath and then to the case, the sheath material shares part of the load. This sheath stress is found to be 266 MPa (39 ksi).

Table II. TIBER TF-coil pack parameters.

Coil cross section (straight leg)		0.143 m^2
Winding pack cross section		0.102 m^2
Number of turns		216
Pack current density		40 A mm^{-2}
Effective area		(21.67 mm)^2
% steel	25	
% insulator	17	
% conductor	34	
% helium	24	

The centering force of the TF coils causes a compressive stress, which is felt by both the pusher coil and the outermost layer of the TF winding. The stress has a maximum on the midplane given by

$$\sigma_c = \frac{B^2_{max}}{2\mu_0} \frac{\frac{8x^2}{3} - 4x + \frac{4}{3x}}{(1 - x^2)^2} ,$$

where

$$x = R_1/R_2 .$$

Note that the stress has the form of the magnetic pressure $B^2_{max}/2\mu_0$ at the inside of the straight leg times a factor (>1 for any real case) to account for the finite thickness of the winding pack. For our case

$$\sigma_c = 53 \text{ MPa } (7.7 \text{ ksi}) .$$

Empirical evidence (Becker/MIT) indicates that this stress can be supported in the winding pack. Special attention will need to be given to supporting the centering loads with the pusher coil.

Nuclear heating of 43 kW in the TF coils must be removed by flowing the helium through the conductors. Because of the competing effects of heat removal and heat generation by friction, there is a minimum temperature increase in the flow path as flow rate is increased. By pancake-winding the coils, two in hand, and by injecting flow on the inner layer, there are 36 flow paths per coil. With an inner conductor nuclear heating rate of 8.1 W/cm^3, a flow rate of 6 kg/s is required with inlet conditions of 6-atm pressure and 4.5 K and outlet conditions of 2 atm and 6.0 K. These parameters represent the maximum refrigeration requirements since they result from an attempt to obtain the minimum helium temperature rise. If greater temperature rise can be tolerated, less refrigeration power will be required.

The pack current density of 40 A/mm^2 corresponds to a conductor current density of 118 A/mm^2. The Nb_3Sn tin conductor can sustain 200 A/mm^2 so that the coil operates at only 60 percent of the critical current. The current sharing temperature is 7.4 K, resulting in a high stability margin even in the most highly heated part of the coil.

Even in steady state the helium liquefier necessary to support the cryogenic heat load is 50 kW. This is about three times the system already constructed for MFTF. An additional 26,000 liter dewar and heat exchanger would be necessary to precool the helium before entering the coils. After exiting the coils, the helium would expand through a Joule-Thomson valve to refill the storage dewar. Using actual costs from the MFTF system, the cost of the helium liquefier and transfer system for TIBER is estimated to cost 18.6 million dollars and to be well within the state of the art.

FUTURE WORK

Further variations in the design parameters are expected before project completion in September 1985; however, our preliminary calculations support the concept that a small (less expensive) superconducting ignition tokamak can be configured by reducing the neutron shielding, and by using both high-magnet current densities and plasma shaping.

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